

James A. FitzPatrick
Nuclear Power Plant
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Michael J. Colomb
Site Executive Officer

November 15, 1999
JAFP-99-0304

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Station P1-137
Washington, D.C. 20555

Subject: **Docket No. 50-333**
LICENSEE EVENT REPORT: LER-99-010 (DER-99-02065)

**Main Turbine Trip and Reactor Scram Due to Degraded Cable in Main
Generator Anti-Motoring Circuit**

Dear Sir:

This report is submitted in accordance with 10 CFR 50.73 (a) (2) (iii).

There are no commitments contained in this report.

Questions concerning this report may be addressed to Mr. Mark Abramski at (315) 349-6305 .

Very truly yours,

A handwritten signature in black ink, appearing to read 'M. Colomb'.

MICHAEL J. COLOMB

MJC:MA:las
Enclosure

cc: USNRC, Region 1
USNRC, Project Directorate
USNRC Resident Inspector
INPO Records Center

IE22

POB 11000

Estimated burden per response to comply with this mandatory information collection request: 50 hrs. Reported lessons learned are incorporated into the licensing process and fed back to industry. Forward comments regarding burden estimate to the Records Management Branch (T-6 F33), U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, and to the Paperwork Reduction Project (3150-0104), Office of Management and Budget, Washington, DC 20503. If an information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not

LICENSEE EVENT REPORT (LER)

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FACILITY NAME (1)

James A. FitzPatrick Nuclear Power Plant

DOCKET NUMBER (2)

05000333

PAGE (3)

1 OF 5

TITLE (4)

Main Turbine Trip and Reactor Scram Due to Degraded Cable in Main Generator Anti-Motoring Circuit

EVENT DATE (5)			LER NUMBER (6)			REPORT DATE (7)			OTHER FACILITIES INVOLVED (8)	
MONTH	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	MONTH	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
10	14	99	99	010	00	11	15	99	N/A	05000
									FACILITY NAME	DOCKET NUMBER
									N/A	05000

OPERATING MODE (9) N

POWER LEVEL (10) 100

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR 5: (Check one or more) (11)

20.2201(b)	20.2203(a)(2)(v)	50.73(a)(2)(i)	50.73(a)(2)(viii)
20.2203(a)(1)	20.2203(a)(3)(i)	50.73(a)(2)(ii)	50.73(a)(2)(x)
20.2203(a)(2)(i)	20.2203(a)(3)(ii)	X 50.73(a)(2)(iii)	73.71
20.2203(a)(2)(ii)	20.2203(a)(4)	50.73(a)(2)(iv)	OTHER
20.2203(a)(2)(iii)	50.36(c)(1)	50.73(a)(2)(v)	Specify in Abstract below or in NRC Form 366A
20.2203(a)(2)(iv)	50.36(c)(2)	50.73(a)(2)(vi)	

LICENSEE CONTACT FOR THIS LER (12)

NAME

Mark Abramski, Sr. Licensing Engineer

TELEPHONE NUMBER (Include Area Code)

315-349-6305

CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIC	CAUSE	SYSTEM	COMPONENT	MANUFACTURER	REPORTABLE TO EPIC

SUPPLEMENTAL REPORT EXPECTED (14)

YES (If yes, complete EXPECTED SUBMISSION DATE):	X NO	EXPECTED SUBMISSION DATE (15)	MONTH	DAY	YEAR

ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines) (16)

On October 14, 1999 at 17:44, an automatic Main Turbine trip followed by a Reactor Scram occurred due to a short circuit between conductors in a control cable in the Main Generator anti-motoring protective relay circuit. (Figure 1). The plant was at 100% power at the time of the Main Turbine Trip. During the event the HPCI system tripped on overspeed. Corrective actions include repairs of the Main Generator anti-motoring circuit and repairs to the HPCI system.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
James A. FitzPatrick Nuclear Power Plant	05000333	99	010	00	2 OF 5

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

EIIIS Codes in []

Event Description

At the time of this event the plant was at 100% power and no evolutions relevant to the event were in progress.

On October 14, 1999 at 17:44, an automatic Main Turbine [TA] trip occurred due to a short circuit between conductors in a control cable in the Main Generator [TB] anti-motoring protective relay circuit. (Figure 1). A Reactor scram occurred as a consequence of the Main Turbine trip based on Turbine Control Valve Fast Closure signals.

Immediately following the Main Turbine trip/Reactor scram, operators entered Abnormal Operating Procedure (AOP) 1, Reactor Scram and Emergency Operating Procedure (EOP) 2, RPV Control and took manual control of the turbine driven Reactor Feed Pumps [SK] (RFPs) to mitigate the anticipated Reactor water level transient which occurs due to shrink. The maximum and minimum Reactor levels and pressures experienced during this event were:

	<u>Minimum</u>	<u>Maximum</u>
Water Level (Inches above Top of Active Fuel)	126.96	238.1
Pressure (psig)	897.7	1112.99

A Group II, Primary Containment [JM] isolation, Reactor Water Clean-up (RWCU) [CE] isolation and Reactor Building [NG] isolation occurred as a consequence of Reactor water level falling below 177 inches above the Top of Active Fuel (TAP).

The High Pressure Coolant Injection (HPCI) [BJ] and Reactor Core Isolation Cooling (RCIC) [BN] systems received initiation signals as a consequence of the Reactor water level transient. Post-transient analysis confirmed that the RCIC system injected for a short time period.

Post-transient analysis also determined that the HPCI system started but the HPCI turbine tripped on overspeed. The HPCI turbine immediately reset from the overspeed condition but by this time, Reactor level had recovered and the HPCI system tripped on high Reactor level per design. The Control Rod Drive (CRD) [AA] system continued to inject for the duration of the transient. The Main Condenser [SG] and Condensate/Condensate Booster Pumps were also available as a heat sink and injection source, respectively for the duration of the event.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
James A. FitzPatrick Nuclear Power Plant	05000333	99	010	00	3 OF 5

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Event Description (cont'd.)

Control Rod position verification following the scram was hindered by lack of "full-in" indication for approximately ten Control Rods on the full core display. EOP-2, RPV Control was therefore exited and EOP-3, Failure to Scram was entered. Control Rod position was verified by a combination of changing out several light bulbs, observing several rods move to a visible "00" position on the four-rod display and in one case, giving the rod an additional insert signal to activate the green "full-in" light. All rods were verified to be full-in within 8-10 minutes after the scram.

While in EOP-3, Failure to Scram, the Control Room Supervisor directed Alternate Rod Insertion (ARI) [JC] initiation, and Core Spray [BM] terminate/prevent and Automatic Depressurization System (ADS) [JE] to be overridden. After all rods were confirmed to be full-in, EOP-3 was exited and EOP-2 entered. After a stabilization period, a Reactor cooldown was commenced and EOP-2 was exited at 2024.

During the cooldown period, it was noted that Main Condenser vacuum began to degrade. The Air Ejector [SH] line up was adjusted to maintain the condenser available as a heat sink. Post transient analysis determined that the degraded condenser vacuum was caused by a failed open vapor suction valve on one of the second stage air ejector jets. This caused short-cycling of non-condensable gas flow when that air ejector jet was not in service.

Cause of Event

The cause of this event was damaged insulation on a control power cable to the Main Generator anti-motoring circuit. The mechanism that caused the damaged insulation has not been identified (Cause Code X). Analysis to determine the cause of the insulation degradation is scheduled for the cycle 15 refuel outage (Fall 2000).

The extent of this condition is limited to this single cable. The insulation resistance of all the conductors in all of the cables run in this common conduit was checked using testing appropriate to the conductor's application (resistance checks or megger checks). These checks verified adequate insulation resistance for all conductors.

Analysis

This event is bounded by the Normal Turbine Trip with Bypass as described in the James A. FitzPatrick Updated Final Safety Analysis Report (UFSAR). Plant safety systems responded as expected to this event with the exceptions noted earlier in this report.

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
James A. FitzPatrick Nuclear Power Plant	05000333	99	010	00	4 OF 5

TEXT If more space is required, use additional copies of NRC Form 366A (17)

Analysis (cont'd.)

The safety significance of the plant trip and failure of the HPCI system was mitigated because the ADS, Residual Heat Removal (RHR) [BO], and Core Spray systems were available as redundant emergency core cooling systems during the duration of the event. The RCIC system was available as a redundant source of high-pressure injection and the Main Condenser and Condensate/Condensate Booster Pumps were available as a heat sink and injection source for the duration of the event.

The HPCI trip on overspeed constituted a Safety System Functional Failure as defined by NEI 99-02 Draft C. Post transient analysis determined that the HPCI discharge piping and some associated instrumentation inboard of the HPCI injection valve were subjected to pressure beyond design as a consequence of the HPCI overspeed condition. The stress on the effected piping was analyzed to determine if the overpressure condition had an adverse effect on the integrity of the pipe. This analysis concluded the piping was not adversely effected and remained operable. The instrumentation subjected to the overpressure condition was either replaced or evaluated for operability. In some cases, replacing instrumentation was deemed to be a more effective use of resources than analyzing the instrumentation for operability.

Corrective Actions

1. The Main Generator anti-motoring circuit was repaired. The repair used spare conductors in the control power cable with the degraded insulation. These spare conductors were verified to have adequate insulation integrity by meg-ohm testing prior to conducting the repair. (Complete)
2. The HPCI System piping was analyzed to determine if the overpressure condition had an adverse effect on the integrity of the pipe. This analysis concluded the piping was not adversely effected and remained operable. (Complete)
3. Perform analysis to determine the cause of the degraded cable. (Scheduled to be complete prior to restart from R014)
4. Replace the degraded cable with a new cable. (Scheduled to be complete prior to restart from R014)

LICENSEE EVENT REPORT (LER)
TEXT CONTINUATION

FACILITY NAME (1)	DOCKET (2)	LER NUMBER (6)			PAGE (3)
		YEAR	SEQUENTIAL NUMBER	REVISION NUMBER	
James A. FitzPatrick Nuclear Power Plant	05000333	99	010	00	5 OF 5

TEXT (If more space is required, use additional copies of NRC Form 366A) (17)

Corrective Actions (cont'd.)

5. The instrumentation subjected to the overpressure condition (which resulted from the HPCI overspeed condition) was either replaced or evaluated for operability.
(Complete)
6. The HPCI system was analyzed to determine the cause of the overspeed trip. The most likely cause was determined to be the Remote Servo. The Remote Servo was replaced and the system was declared operable based on surveillance testing required by Technical Specifications.
(Complete)
7. Control Rod position indication problem is a known problem with a comprehensive corrective action plan in place. Elements of this plan include:
- a. A modification to change the light bulbs on the full core display to Light Emitting Diodes. This modification has been installed. This design is not prone to burnout, as was the old incandescent bulb design.
(Complete)
 - b. Installation of a computer based system to record rod positions as they are inserted into the core. (Scheduled to be complete prior to restart from R014)

Additional Information

Similar Events: None

Failed Components: None

Main Generator Anti-Motoring Circuit Simplified Block Diagram

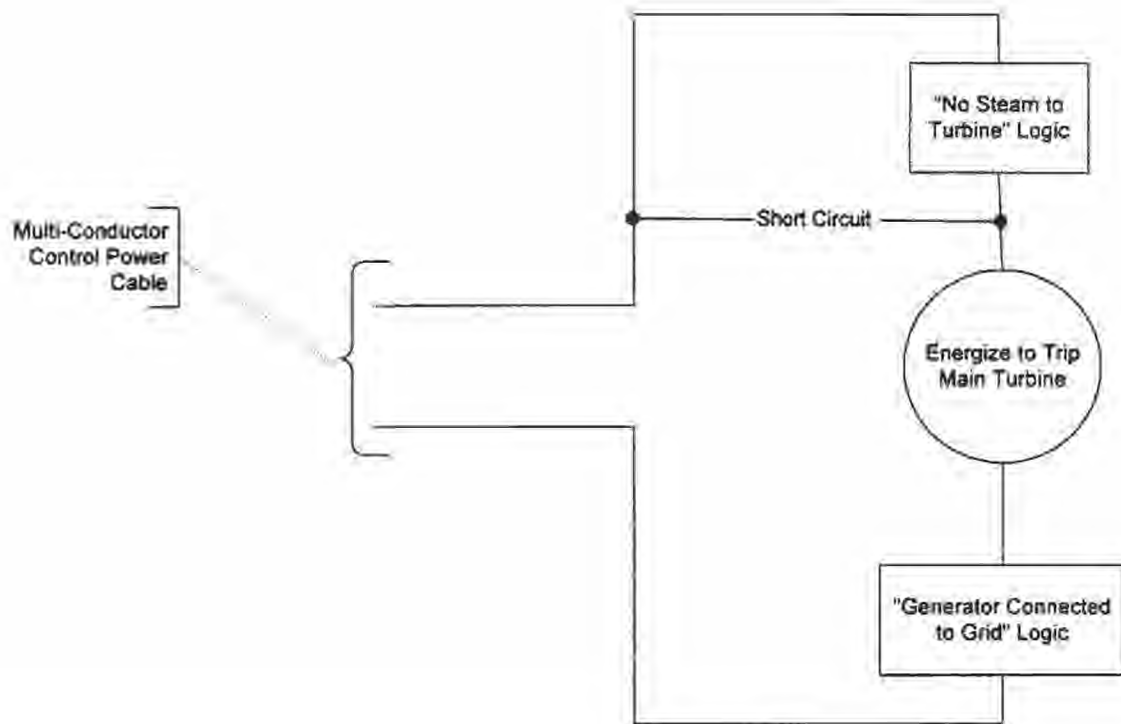


Figure 1

LER 99-010